NON-PUBLIC?: N

ACCESSION #: 8907050489
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Clinton Power Station PAGE: 1 of 7

DOCKET NUMBER: 05000461

TITLE: Failure of the Motor Driven Reactor Feedwater Pump Regulating Valve Results in a Level Transient and the Insertion of a Manual Scram Signal
EVENT DATE: 05/26/89 LER #: 89-022-00 REPORT DATE:

OPERATING MODE: 1 POWER LEVEL: 019

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D. R. Morris, Director, Plant Operations TELEPHONE: (217) 935-8881 - extension 3205

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: EJ COMPONENT: V MANUFACTURER: C600
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

This report documents the insertion of a manual scram signal on May 26, 1989 and again on June 1, 1989. On May 26, 1989, reactor water level began dropping. The motor driven reactor feedwater pump (MDRFP) regulating valve 1FW004 did not open in response to the decreasing level. When indicated reactor water level dropped to approximately ten inches, a manual scram signal was inserted. The failure of valve 1FW004 was determined to be caused by a faulty servo-amplifier board. The board was replaced. The plant entered Mode 2 (STARTUP) on May 27, 1989. On June 1, 1989, power was being reduced in order to shut the reactor down to repair the seal on Reactor Recirculation pump B. After switching feedwater control from the turbine driven reactor feedwater pump to the MDRFP, valve 1FW004 failed to shut in response to its control signal. Reactor water level began increasing. When indicated reactor water level reached forty-five inches, a manual scram signal was inserted.
The cause of these events was the failure of valve 1FW004 to respond to its control signal. Troubleshooting identified two probable causes: vibrations

caused loose electrical connections on the servo-amplifier board; silicon-controlled rectifiers on the board had failed. The board has been replaced and relocated.

END OF ABSTRACT

DESCRIPTION OF EVENT

On May 26, 1989, at approximately 0430 hours, the plant was in Mode 1 (POWER OPERATION). In preparation for starting the turbine generator, reactor RCT! power was raised to nineteen percent. Reactor water level was at thirty-one inches as indicated on the narrow range reactor water level indicator LI). (Note: The reference point for reactor water level is 162 inches above the top of active fuel. This is zero inches as indicated by instrumentation.) Feedwater was being provided by the motor MO) driven reactor feedwater EJ! pump (MDRFP) P!.

At approximately 0435 hours, the indicated reactor water level dropped to thirty and eight-tenths inches, the Reactor Recirculation System (RR) AD! flow control valve FCV! began closing, as designed, and reactor power decreased to approximately twelve percent. Reactor water level continued to drop. An operator was sent to investigate the operation of MDRFP regulating valve V!, 1FW004. The control switch for valve 1FW004 was in the automatic position and the startup level controller was controlling the operation of the valve. The operator verified that valve 1FW004 was being fed a 100 percent feedwater demand signal, but that the valve was not fully open.

The operator attempted to reposition valve 1FW004 by switching its control switch back and forth from automatic to manual a number of times, and by changing control of the valve from the startup level controller to the loop controllers. The valve did not reposition as a result of these actions. When the indicated reactor water level had dropped to approximately ten inches, the Line Assistant Shift Supervisor, in anticipation of an automatic reactor scram at an indicated reactor water level of eight inches, ordered the control room operator to initiate a manual scram.

At 0441 hours, the control room operator placed the mode switch in the shutdown position, which inserted a manual scram signal. All control rods were inserted into the reactor. The indicated reactor water level dropped to negative nine and two-tenths inches. In order to restore reactor water level, operators started condensate and condensate booster (CB) KA! pumps. Reactor water level began increasing and reached an indicated level of approximately forty-five inches. As expected, reactor pressure began decreasing as water was injected into the reactor.

Operators took a number of actions to maintain reactor pressure in order to minimize the reactor cooldown rate. When pressure dropped to approximately 600 pounds per square inch (psi) the MDRFP was removed from service. Pressure continued to drop, and when it had dropped to approximately 550 psi, steam to the Offgas System (OG) WF! was isolated. Operators then isolated OG and as a result, the steam jet air ejectors (SJAE) EJR! were removed from service.

Condensate polishers (CP) KD! G and J were removed from service, as were condensate booster pump A and condensate (CD) SD! pump B. The motor driven condenser COND! vacuum pumps were started in order to maintain a vacuum in the condenser. The feedwater shutoff valves, 1B21-F065A and B, were shut.

In order to maintain reactor level while reactor pressure was decreasing, the Reactor Water Cleanup system (RWGU) CE! was lined-up to reject water from the reactor to the condenser.

Plant conditions stabilized at approximately 0500 hours.

After plant conditions were stabilized, an investigation was initiated to determine the cause of the failure of valve 1FW004. Data recorded during the event indicated that feedwater flo
was not consistent with the control
signals being fed to valve 1FW004. This inconsistency indicated that the position of valve 1FW004 was not changing in response to its control signals.
In an attempt to repair this malfunction, on May 27, 1989, the servo-amplifier (AMP) board for valve 1FW004 was replaced in accordance with Maintenance Work Request (MWR) D14075. Following replacement of the servo-amplifier board, testing verified that the valve was responding correctly to inserted control signals.

Since the valve failure could not be attributed to any other cause, following management review of the event and of the completed corrective actions, a decision was made to return the plant to power operation.

On May 27, 1989, at 1325 hours, the plant went from Mode 3 (HOT SHUTDOWN) to Mode 2 (STARTUP). Control rods were withdrawn and the reactor became critical at 2311 hours.

On May 28, 1989, at 1446 hours, the plant entered Mode 1.

On June 1, 1989, the plant was in Mode 1 and the reactor was at forty-three percent power. At approximately 0044 hours, RR pump A was transferred from slow to fast speed. At approximately 0045 hours, RR pump B was transferred from slow to fast speed. Almost immediately the pump seal pressure rapidly decreased causing alarms AI14) to annunciate in the main control room. At 0055 hours, flow to the drywell floor drain (DRN) exceeded five gallons per minute (gpm). In accordance with the Emergency Plan, because of the high flow to the drywell floor drain from an "unknown" source, the Shift Supervisor declared an Unusual Event. At 0056 hours, operators noted that drywell pressure was increasing and that flow to the drywell floor drain from the leaking RR pump B seal had reached sixty-four gpm. Operators then shifted the RR pumps to slow speed.

At 0057 hours, operators shut the isolation valves for RR loop B. At 0058 hours, operators identified that drywell cooling system (VP) VB! chiller CHU! 'A' had tripped due to low chiller condenser water pressure, Operators isolated rod drive seal supply to RR pump B at this time.

At 0100 hours, the Shift Supervisor upgraded the emergency to an Alert due to the flow to the drywell floor drain exceeding fifty gpm.

Although leakage from the pump seal was eliminated when RR loop B was isolated, drywell pressure continued to increase because VP chiller A had tripped. At 0105 hours, drywell pressure reached 1.36 psi gauge (psig). Since VP chiller A had tripped and since drywell pressure was increasing, in accordance with procedures, operators started the Combustible Gas, Control System (CGCS) BB! compressors COMP! to decrease drywell pressure. At 0110 hours, when drywell pressure had dropped to 1.02 psig, operators started VP chiller B. After confirming that drywell pressure was continuing to decrease, operators removed the CGCS compressors from service.

At 0140 hours, the emergency was downgraded to an Unusual Event due to the reduction of the flow to the drywell floor drain to less than one gpm following isolation of RR loop B.

At 0140 hours, after stabilizing plant conditions, operators began inserting control rods in order to shut down the reactor. The reactor was being shut down to investigate the failure of, and to repair, the RR pump B seal.

By 0325 hours, plant conditions had stabilized, flow to the drywell floor drain was less than one gpm, and radiation surveys indicated that no detectable radiological release was made to the environment as a result of the event. Based on these facts, the Shift Supervisor secured from the Unusual Event.

At 0446 hours, the control room operator placed the mode switch in the startup/hot standby position and the plant entered Mode 2.

At 0555 hours, the reactor was at two percent power. Operators shifted feedwater flow supply from the turbine driven reactor feedwater pump (TDRFP) to the MDRFP. Valve 1FW004 was in automatic being controlled by the startup level controller. After placing the MDRFP in service, reactor water level began to increase. In order to control reactor water level, in accordance with procedures, operators began throttling feedwater shutoff valve 1B21-FO65B shut. Throttling valve 1B21-FO65B shut caused reactor water level to decrease. In response to the decreasing reactor water level, valve 1FW004 began to open as expected. Reactor water level again began to increase, however valve 1FW004 would not shut in response to its control signal. Operators attempted to reposition valve 1FW004 by

switching its control switch back and forth from automatic to manual a number of times, and by changing control of the valve from the startup level controller to the loop controller. The valve did not reposition as a result of these actions. Operators also attempted to control reactor water level by throttling 1B21-FO65B shut.

When indicated reactor level reached forty-five inches, in anticipation of an automatic reactor scram at an indicated reactor water level of fifty-two inches, the control room operator placed the mode switch in the shutdown position which inserted a manual scram signal.

All control rods were inserted. The inboard Main Steam Isolation Valves were shut in order to minimize the reactor cooldown rate. CD pump A, CB pump B, and condensate polisher J were removed from service.

Plant conditions were stabilized at approximately 0700 hours.

The plant entered Mode 4 (COLD SHUTDOWN) at 1123 hours on June 1, 1989.

CAUSE OF EVENT AND CORRECTIVE ACTIONS

The cause of this event is attributed to an equipment failure. Valve 1FW004 failed to respond to control signals, both manual and automatic, causing reactor level transients.

In order to determine the cause of, and corrective actions for, the failure of valve 1FW004, a troubleshooting plan was developed. Actions performed in accordance with the troubleshooting plan systematically isolated and assessed the performance of the electric/electronic and hydraulic operation of valve 1FW004 as well as the mechanical motion/response of the valve.

Troubleshooting identified a number of discrepancies in the electric/electronic operation of valve 1FW004. Two of the discrepancies, loose electrical connections and faulty servo-amplifier boards, were determined to be the factors responsible for the failure of valve 1FW004 to respond to changes in its control signal.

Intermittent contact symptoms were observed during inspection and electrical testing of the wiring and terminations. When Control and Instrumentation (C&I) technicians moved the connecting wire for the servo-amplifier board while investigating the failure of valve 1FW004 to respond to an inserted control signal, the valve responded. This symptom was exhibited repeatedly and is attributable to possible loose connections on the servo-amplifier board. The most probable cause of the loose connections is vibration of the servo-amplifier board. In order to reduce vibration of the servo-amplifier board, the board was moved from its original location (mounted on valve 1FW004) and mounted on a wall in the vicinity of the valve.

Additional corrective actions included testing of the servo-amplifier board. This testing was independent of the electrical connection testing. The servo-amplifier board which was installed on May 27, 1989, following the manual reactor scram of May 26, 1989, was determined to be faulty. Two additional boards were installed and tested. One was found to be faulty and the other is believed to have been damaged during testing following installation. Two of the four boards installed on valve 1FW004 were factory tested subsequent to installation. The factory testing determined that the silicon-controlled rectifiers (SCRs) RECT! in the board had failed. As a result, the servo-amplifier board was not providing full voltage rectification to the solenoid valve in response to control signals. Therefore, motion of the valve was not consistent with control signals.

The actuator for valve 1FW004 was manufactured by the Paul Monroe Corporation. This actuator has a unique electric/hydraulic control and its sole application at Clinton Power Station is on valve 1FW004.

A new servo-amplifier board has been installed and repeated testing has confirmed that it operates properly.

A "dry-run" test and a functional test of valve 1FW004 have been completed with satisfactory results.

Valve 1FW004 will be electronically monitored at periodic intervals during the in-progress power ascension. If control problems are detected with the operation of valve 1FW004 the plant will be placed in a safe condition to facilitate repairs.

As a long term corrective action, Illinois Power Company is evaluating the feasibility of a plant modification to replace valve 1FW004 and/or its control system. This evaluation is expected to be completed by November 15, 1989.

In accordance with action plans developed by the Nuclear Station Engineering Department, both the RR pump seal which led to the initiation of the plant shut down on June 1, 1989, and the VP chiller which tripped during the Unusual Event, have been repaired and tested with satisfactory results.

On June 18, 1989, at 2215 hours, the plant entered Mode 2.

ANALYSIS OF EVENT

These events are reportable under the provisions of 10CFR50.73(a)(2)(iv) due to the actuation of the Reactor Protection System (RPS) JE!. Control room operators inserted a manual scram signal on May 26, 1989 at 0441 hours and on June 1, 1989 at 0610 hours.

These events were not safety significant for existing plant conditions or other plant modes. Chapter 15 of the Clinton Power Station Updated Safety Analysis Report contains an analysis of the effects of manually initiated reactor scrams. The effects have been found to be acceptable.

ADDITIONAL INFORMATION

LER 87-029-00 discusses an automatic RPS actuation caused by failure of valve 1FW004. The valve failure resulted from a faulty solenoid valve and control circuit board.

LER 87-025-00 discusses a manual RPS actuation caused by failure of valve 1FW004. The failure resulted from a loss of hydraulic fluid that occurred when a pressure switch diagram ruptured.

For further information regarding this event, contact D. R. Morris, Director - Plant Operations at (217) 935-8881, extension 3205.

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ILLINOIS POWER COMPANY

CLINTON POWER STATION, P.O. BOX 678, CLINTON, ILLINOIS 61727

June 26, 1989

10CFR50.73

Docket No. 50-461

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 89-022-00

Dear Sir:

Please find enclosed Licensee Event Report No. 89-022-00: Failure of the Motor Driven Reactor Feedwater Pump Regulating Valve Results in a Level Transient and the Insertion of a Manual Scram Signal. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

D. L. Holtzscher
Acting Manager -
Licensing and Safety

TS/krm

Enclosure

cc: NRC Resident Office
NRC Region III, Regional Administrator
INPO Records Center
Illinois Department of Nuclear Safety
NRC Clinton Licensing Project Manager

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